Third United Nations International Conference on the Peaceful Uses of Atomic Energy

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Abstract

THE INDUSTRIAL PRODUCTION OF FUEL ELEMENTS

by MM. BOUSSARD - Chef du Département de Construction des Piles
BOUDOURESQUE - Ingénieur au Département de Métallurgie
DENCOURS - Ingénieur au Département de Métallurgie
TRIEULE - Ingénieur au Département de Construction
LAFAURIE - Ingénieur au Département de Construction

(Commissariat à l'Energie Atomique, France)

The authors deal successively with the industrial production of fuel elements for power reactors of the natural uranium-graphite-gas type, and more particularly for the EDF power stations, and with the industrial production of fuel elements containing enriched uranium designed for swimming-pool type reactors.

1st part : advanced fuel elements for the EDF reactors.

After recalling the characteristics of the fuel elements now being produced industrially for the MARCOULE and CHINON reactors, the authors give the various steps leading to the industrial production of a new type of fuel element both as concerns the can, and in certain cases the graphite sleeve, and the fuel itself.

As for as the production of the fuel is concerned, they describe the various operations, stressing the original aspects of the production and of the equipment such as :

- casting in hot moulds
- thermal treatments of Uranium containing 1% in weight molybdenum

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- welding of the pellets for closing the tubes of uranium
- canning
- controls in the various steps.

As far as can production is concerned they show why the extruded can was replaced by a machined can and give a few characteristics of the equipment used as well as the controls effected. They give also some details concerning the production and machining of the sleeves.

After recalling the state of the nuclear fuel industry in France in mid-1964, the authors stress the economic aspects of the production of fuel elements.

They show the relative importance of capital costs on the cost price of the fuel itself and examine the various items involved. They analyse the cost price of a completed fuel element using present date

In conclusion the authors show the particular points which should be the subject of future efforts in order to decrease the cost of a production which is perhaps delicate but now will defined, and review the development of this new industrial branch.

2nd part : industrial production of fuel elements for swimming-pool type reactors.

The authors show how the problem of the industrial production of rolled fuel elements has been solved in France, and give the three steps involved :

1 - Assembly of the plates made in the U. S. A.

2 - Rolling of the cores made in the U. S. A. to obtain the plates.

3 - Fabrication of the U-Al alloy and production of the cores.

They then recall briefly the characteristics of the different fuel elements now in production.

A description is given of the various stages of the production including information about the equipment ; stress is laid on the extent of the controls carried out at each stage.

In conclusion the authors consider the future development of this type of production taking into account the improvements planned and those which are possible.

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Abstract

STUDIES AND MANUFACTURE OF PLUTONIUM FUEL

by : Plutonium Department

(Commissariat à l'Energie Atomique, France)

The studies carried out at the C. E. A. on the properties of fast neutron reactor fuels, the manufacture of fuel elements and their behaviour under irradiation are broadly outlined.

The metal fuels studied are the ternary alloys U Pu Mo, U Pu Nb, U Pu Ti, U Pu Zr, the ceramic fuels being mixed uranium and plutonium oxides, carbides and nitrides obtained by sintering.

Results are given on the manufacture of uranium fuel elements containing a small proportion of plutonium, used in a critical experiment, and on the first experiments in the manufacture of fuel elements for the reactor Rapsodie.

Finally the results of irradiation tests carried out on the prototype fuel pins for Rapsodie are described.

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Abstract

RECENT DEVELOPMENTS IN THE FIELD OF REFRACTORY FUELS

by A. ACCARY and R. DELMAS

(Commissariat à l'Energie Atomique, France)

The main part of the work carried out in the field of ceramic fuels by the Commissariat à l'Energie Atomique during recent years, has been in the direction of uranium dioxide and the uranium-carbon alloys.

Uranium dioxide is being studied with the aim of using it as a fuel in the first core of EL-4, in which an integrated thermal conductivity of 29 W/cm is expected at the hottest point for a surface temperature of about 750°C.

We concentrated on developing a process for preparing a dioxide powder of suitable characteristics and for sintering this powder, and on evaluating the main properties of the material obtained in the light of the conditions under which they will be used :

- micro-structural aspect and pore distribution,
- mechanical and thermal behaviour in cylindrical form,
- control of excess oxygen in the industrial products,
- behaviour of the gaseous fission products at high temperatures after or during the course of irradiation.

Our aim in the case of uranium carbides has been to determine the conditions of industrial manufacturing of a suitable fuel with a composition close to UC.

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This has led us to undertake a number of fundamental investigations into :

- the domain of existance of non-stoichiometric UC,
- the influence of elements of O and N on the properties of the UC in which they are dissolved,
- the compatibility of uranium-carbon alloys with the different metallic or ceramic materials used for the sheath,
- the corrosion of uranium-carbon alloys by H_2O and CO_2,
- methods of preparing high purity samples,
- in-pile irradiation devices for the investigation of these materials in the region of possible operating temperatures.

In parallel with these fundamental investigations, we have attempted to define a proceedure for the fabrication of uranium-carbon alloys of composition very close to UC which would, in its industrial application, lead to better results than the existant methods from the technical or economical points of view (i.e. sintering or arc fusion). This latter method is based on fusion by electronic bombardment associated with the continuous casting.

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Abstract

RECENT DEVELOPMENTS CONCERNING FRENCH FUEL ELEMENTS USED IN NATURAL URANIUM - GRAPHITE - CO_2 REACTOR SYSTEMS

by MM. M. SALESSE
J. A. STOHR
G. JEANPIERRE

(Commissariat à l'Energie Atomique, France)

The policy followed in France for the development of fuel elements for reactors belonging to the Electricité de France has been to benefit as much as possible, for each new pile from the most recent technical progress by developing in each case a fuel element allowing the maximum power per channel.

The two latest fuel elements thus studied by the French Atomic Energy Commission are of two different types :

- a tubular uranium element closed at both ends and cooled externally.

This type of element, chosen for the reactors EDF 2, EDF 3 and EDF 4 makes it possible to attain maximum specific powers of the order of 6 MW/metric ton.

An open tubular uranium element cooled both internally and externally, called an "annular element" which in being studied as a possibility for EDF 5. Such an element makes it possible to attain specific powers of over 12 MW/metric ton.

The two types of element have the following common characteristics
- the can, for external cooling, has herning-bone type fins.

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This type of profile which has been vastly improved recently thereby increasing its thermal efficiency, has the important advantage of avoiding vibration of the element, but has posed problems of resistance to thermal cycling necessitating much research.

- the fuel rods are placed inside graphite jackets, this limiting the vertical forces to which they are subjected and protecting them during charging and discharging.

On the other hand, these elements present very different problems as for as the following points are concerned :

- the characteristics required of the uranium tubes apart of course from a good dimensional stability during irradiation in the two cases are in the case of the closed tubes a very high resistance to external pressure, and in the case of the annular elements a low neutron absorption. Thus for each of these two cases it has been necessary to develop a suitable type of alloy.

- a possible loosening of the can during thermal cycling, which is peculiar to the internal can of the annular element, has necessitated very much research work.

- the exact temperature drop at the contact between the uranium and the can, and the strength of the lower end of the cartridge are points which are increasingly crucial in the case of the annular element.

All in all the annular element thus calls for a great research effort. This effort is justified by the big step forward in which it will result in the case of the EDF reactors thanks to its high specific power and to the high weight of uranium in each cartridge.

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Abstract

TRANSFORMATIONS OF HIGHLY ENRICHED URANIUM INTO METAL OR OXIDE

DESIGN AND ENGINEERING OF WORKSHOPS

(Commissariat à l'Energie Atomique, France)

THE ENRICHED URANIUM WORKSHOPS in CADARACHE have a double purpose : on the one hand to convert uranium hexafluoride into metal or oxide, and on the other hand to recover the uranium contained in scrap materials produced in the different metallurgical transformations.

The principles that have been adopted for the design and safety of these workshops are reported. The nuclear safety is based on the geometrical limitations of the processing vessels.

To establish the processes and the technology of these workshops, many studies have been made since 1960, some of which have led to original achievements.

The uranium hexafluoride of high isotopic enrichment is converted either by injection of the gas into ammonia or by an original process of direct hydrogen reduction to uranium tetrafluoride.

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The uranium contained in uranium-zirconium metal scrap can be recovered by combustion with hydrogen chloride followed treatment of the uranium chloride by fluorine in order to obtain the uranium in the hexafluoride state.

Recovery of the uranium contained in various scrap materials is obtained by a conventional refining process : combustion of metallic scrap, nitric acid dissolution of the oxide, solvent purification by tributyl phosphate, ammonium diuranate precipitation, calcining, reduction and hydrofluorination into uranium tetrafluoride, bomb reduction by calcium and slag treatment.

Two separate workshops operate along these lines : one takes care of the uranium with an isotopic enrichment of up to 3 p.100, the other handles the high enrichments.

The handling of each step of this process, bearing in mind the necessity for nuclear safety, has raised some special technological problems and has led to the conception of new apparatus, in particular the roasting furnace for metal turnings, the nitric acid dissolution unit, the continuous precipitator and eversafe filter and dryer for ammonium diuranate, the reduction and hydrofluorination furnace and the slag recovery apparatus. These are described successively, together with the studies which led to their development.

The civil engineering construction was begun in june 1962. The workshops are to start production in may 1964.

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Abstract

BEHAVIOUR OF URANIUM UNDER IRRADIATION

by : Metallurgy Department - Plutonium Department

(Commissariat à l'Energie Atomique, France)

The main results obtained in a study of the formation of defects caused in uranium by fission at low temperature are reported. By irradiation at 20°K it was possible to determine the number of Frenkel pairs produced by one fission. An analysis of the curves giving the variations in electrical resistivity shows the size of the displacement spikes and the mechanism of defect creation due to fission. Irradiations at 77°K gave additional information, showing behaviour differences in the case of recrystallised and of cold worked uranium.

The diffusion of rare gases was studied using metal-rare gas alloys obtained by electrical discharge, and samples of irradiated uranium. Simple diffusion is only responsible for the release of the rare gases under vacuum in cases where the rare gas content is very low (very slightly irradiated U). On the other hand when the concentration is higher (samples prepared by electrical discharge) the gas is given off by the formation, growth and coalescence of bubbles ; the apparent diffusion coefficient is then quite different from the true coefficient and cannot be used in calculations on swelling.

The various factors governing the phenomenon of simple diffusion were examined. It was shown in particular that a small addition of molybdenum could reduce the diffusion coefficient by a factor of 100.

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The precipitation of gas in uranium (Kr), in silver (Kr) and in Al-Li alloy (He) have been followed by measurement of the crystal parameter and of the electrical resistivity, and by electron microscope examination of thin films. The important part played by dislocations in the generation and growth of bubbles has been demonstrated, and it has been shown also that precipitation of bubbles on the dislocation lattice could block the development of recrystallisation.

The results of these studies were compared with observations made on the swelling of uranium and uranium alloys U Mo and U Nb strongly irradiated between 400 and 700°C.

In the case of Cubic phase alloys (U Nb, U Mo) the size and distribution of the bubbles are greatly influenced by the existance of a polygonisation lattice, formed before irradiation, inside the grains. During high temperature annealings the bubbles situated on the polygonisation lattice grow much faster than those which are distributed inside the crystal.

In the case of uranium the bubbles are either distributed at random, or grouped in flat accumulations which can give rise to transgranular cracks, according to the state of the metal and the irradiation temperature. At high temperature an intergranular decohesion is also observed.

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Abstract

COMPATIBILITY PROBLEMS OF CANNING MATERIALS WITH CARBON DIOXIDE AT HIGH TEMPERATURES

(Commissariat à l'Energie Atomique, France)

by DARRAS R. and LORIERS H.

The adoption in France of carbon dioxide under pressure as a heat-carrying fluid in "advanced" reactors of the gas-graphite and gas heavy water types has led to the necessity of finding a canning material capable of replacing magnesium alloys. Actually these latter can no longer be used above about 500°C, because of the proximity to their melting points and of their mechanical properties which become under these conditions insufficient, although their oxidation resistance in the presence of carbon dioxide is still acceptable.

Beryllium which is particularly attractive because of its low neutron capture cross-section, has a very big disadvantage, amongst others : its use in the presence of carbon dioxide et 600°C is only possible if the gas is perfectly dry, the water-vapour partial pressure being the determining factor calling for a degree of drying which increases with increasing absolute pressure. In the opposite case, after a short incubation period, the oxidation accelerates and leads to an intergranular corrosion which is rapidly destructive. Nevertheless, beryllium-calcium or beryllium-magnesium alloys containing 0,5 p. 100 of the addition element make it possible to overcome this difficulty ; they may be used in the presence of a few hundred vpm of water vapour up to at least 700°C.

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The metallurgical problems convected with the applicability of beryllium or its alloys have led however to the consideration of possibly using provisionally austenitic stainless steels.

These materials are intrinsically very resistant to oxidation, but, as only small thicknesses can be used because of their high capture cross-section, it is necessary to choose the grades which are least prone to oxidation. Above 600°C, the niobium stabilised grades, without addition of molybdenum, are the most acceptable. Above 700°C, the 20 Cr-25 Ni + Nb grade appears more suitable than the conventional 18 Cr - 10 Ni + Nb grade, especially as it is less liable to local oxide penetrations.

Furthemore, iron-aluminium alloy containing 25 p. 100 by weight of aluminium has a remarkable oxidation resistance while at the same time it allows an appreciable saving of neutrons ; its production however is still difficult technologically.

Finally recent research has showed that zirconium based alloys containing 1 to 4 p. 100 of copper by weight could become an acceptable solution up to temperatures of about 600°C. These alloys are in fact oxidised much slower than unalloyed zirconium and other conventional alloys which had previously been eliminated for this reason.

In conclusion, amongst the materials capable of resisting to the physico-chemical conditions to which would be subjected the fuel element cans in "medium temperature" reactors, the most suitable for immediate application would appear to be austenitic stainless steels and zirconium - copper alloys since the metallurgy of both is comparatively well known. In the future the practical advantages of beryllium and iron-aluminium alloys should increase as a result of the research carried out with a view to improving their technological and mechanical properties.

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Abstract

COST OF TRANSPORTING IRRADIATED FUELS AND MAINTENANCE COSTS OF A CHEMICAL TREATMENT PLANT FOR IRRADIATED FUELS

by Y. SOUSSELIER

(Commissariat à l'Energie Atomique, France)

Numerous studies have been made of the cost of a fuel cycle, but many of them are based on a priori studies and are therefoce to be treated with reserve. Thus, in the part dealing with the treatment of irradiated fuels, some important factors in the cost have only rarely been given on the basis of practical experience : the cost of transporting the fuels themselves and the plant maintenance costs.

Investigations relating to transport costs are generally based on calculations made from somewhat arbitrary data. The studies carried out in France on the transport of irradiated uranium between the EDF reactors at Chinon and the retreatment plant at La Hague of the irradiated uranium from research reactors to foreign retreatment plants, are reported ; they show that by a suitable choice of transport containers and details of expedition it has been possible to reduce the costs very considerably. This has been achieved either by combining rail and road transport or by increasing the writ capacities of the transport containers : an example is given of a container for swimming-pool pile elements which can transport a complete pile core at one time, thus substantially reducing the cost.

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Studies concerning the maintenance costs of retreatment plants are rarer still, although in direct maintenance plants these figures represent an appreciable fraction of the total treatment cost. An attempt has been made, on the basis of operational experience of a plant, to obtain some idea of these costs. Only maintenance proper has been considered, excluding subsidiary operations such as the final decontamination of apparatus, the burial of contaminated material and radioprotection operations. Maintenance has been divided into three sections : mechanical maintenance, maintenance of electrical equipment and maintenance of control and adjustment apparatus. In each of these sections the distinction has been made between manpower and the material side. In order to allow comparisons to be made with other plants, an attempt has been made to express the maintenance costs as a percentage of the investments corresponding to each of the sections considered.

The unequal aspect of maintenance in a plant of this type is demonstrated, this being due particularly to the inacessibility of most of the equipment during operation.

Finally some conclusions are drawn on what the maintenance costs could be and the means of reducing them in future plants.

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Abstract

THE REPROCESSING OF IRRADIATED FUELS
IMPROVEMENT AND EXTENSION OF THE SOLVENT EXTRACTION PROCESS

by P. FAUGERAS, A. CHESNE, X. TALMONT, R. BONNIAUD

(Commissariat à l'Energie Atomique, France)

Improvements made in the conventional tri-butylphosphate process are described, in particular : the concentration and the purification of plutonium by one extraction cycle using tri-butyl-phosphate with reflux ; and the use of an apparatus working continuously for precipitating plutonium oxalate, for calcining the oxalate, and for fluorinating the oxide.

The modifications proposed for the treatment of irradiated uranium-molybdenum alloys are described, in particular, the dissolution of the fuel, and the concentration of the fission product solutions.

The solvent extraction treatment is used also for the plutonium fuels utilized for the fast breeder reactor (Rapsodie). An outline of the process is presented and discussed, as well as the first experimental results and the plans for a pilot plant having a capacity of 1 kg/day.

The possible use of tri-lauryl-amine in the plutonium purification cycle is now under consideration for the processing plant at La Hague. The flowsheet for this process and its performance are presented.

The possibility of vitrification is considered for the final treatment of the concentrated radioactive wastes from the Marcoule (irradiated uranium) and La Hague (irradiated uranium-molybdenum) Centers. Three possible processes are described and discussed, as well as the results obtained from the operation of the corresponding experimental units using tracers.

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Abstract

THE REPROCESSING OF IRRADIATED FUELS BY HALIDES AND THEIR COMPOUNDS

by M. BOURGEOIS, P. FAUGERAS

(Commissariat à l'Energie Atomique, France)

A brief description is given of the experiments leading to the choice of the process : volatilization of fluorides by gas phase attack.

The chemical process is described for certain current types of clad fuels : the aluminium or the zirconium cladding is first volatilized as chloride by attack with gaseous hydrogen chloride. The uranium is then transformed into volatile hexafluoride by attack with fluorine. These reactions are carried out consecutively in the same reactor in the presence of a fluidized bed of alumina which facilitates heat exchange.

The experiments have been carried out in quantities from 100 gms to several kilograms of fuel, first without activity, and then with tracers.

A description is given of the laboratory research which was carried out simultaneously on the separation of uranium and plutonium fluorides.

Finally, an apparatus is described which is intended to test the process on irradiated fuel at an activity level of several thousands of curies of fission products.

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Abstract

SIX YEARS'WORKING EXPERIENCE OF THE MARCOULE PLANT FOR TREATMENT OF IRRADIATED FUEL

by MM. C. JOUANNAUD
M. CHAMBON
A. FONTAINE
R. RIOLFO

(Commissariat à l'Energie Atomique, France)

The irradiated fuel treatment plant at Marcoule began treating rods from the pile G 1 in July 1958. These six years'experience of the plant in operation have led to the confirmation or revision of the original ideas concerning the process as well as the technology or methods of exploitation.

The process as a whole has suffered little modification, the performances having proved better than originally foreseen ; the only alterations made were justified by greater simplicity of operation, better nuclear security (criticality) or for technological reasons. The processes of plutonium reduction from valency IV to valency III by uranium IV, and of concentration of fission product solutions in the presence of formaldehyde, have always given complete satisfaction.

The initial concept of direct maintenance of the installations has been justified by experience. Certain maintenance jobs, originally considered impossible after the start of operations, have proved feasible and have been car-

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ried out under acceptable conditions ; a number of examples are given. From experience it has been possible to define optimal conditions for the design of these installations such as to provide a maximum in robustness and ease of maintenance. The advantages of continuonsly-operating equipment have been shown.

Certain installations have been altered in accordance with these new ideas.

Analytical checking in the laboratory has been profoundly modified, and the plans adopted are such that complete safety in work on radioactive solutions is compatible with a very good working speed. Experience has also shown the advantages of having a group on the spot to carry out short-term applied studies.

Finally, a strict working discipline and excellent collaboration with the radiation protection service have enabled us to reach the end of these six years, during some of which the exploitation was intensive, without irradiation accident.

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Abstract

SPECIAL GRAPHITES

by MM. P. LEVEQUE - C.E.N. SACLAY
F.M. LANG - C.E.N. SACLAY
J. RAPPENEAU - C.E.N. SACLAY
P. BLUM - C.E.N. GRENOBLE
L. BOCHIROL - C.E.N. GRENOBLE
CORNUAULT - PECHINEY
GALY - PECHINEY

(Commissariat à l'Energie Atomique, France)

A large fraction of the work undertaken jointly by the Commissariat à l'Energie Atomique (C.E.A.) and the Pechiney Company has been the improvement of the properties of nuclear pile graphite and the opening up of new fields of graphite application. New processes for the manufacture of carbons and special graphites have been developed : forged graphite, pyrocarbons, high density graphite, agglomeration of graphite powders by cracking of natural gas, impervious graphites. The physical properties of these products and their reaction with various oxidising gases are described. The first irradiation results are also given.

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Abstract

CONSTRUCTION OF THE CORE OF THE "HEAVY WATER-GAS" REACTOR EL 4

by Jean L. BERNARD
H. FOULQUIER
P. THOME

(Commissariat à l'Energie Atomique, France)

The core of this reactor consists of a vessel containing heavy water, through which pass a sery of pressure tubes for circulation of the cooling gas under boat pressure.

The basic specifications which greatly influenced the design of this construction relate to aspects of safety in operation (fuel loading from both faces of the reactor, replacement of the components on both faces), neutronic demands (minimum absorption of the components lattice parameter, diameter of the pressure tubes) and thermal considerations (output temperature 500°C).

These specifications have led to a horizontal arrangement of the pressure tubes and raised very difficult problems of clearance, which make it impossible (for the dimensions of EL 4) to resort to expansion bellows on the pressure tubes. The result is a semi-rigid vessel in which the pressure tubes contribute to a large extent the mechanical resistance of the system by acting as a brace, whence the high stresses on the joints and pressure tubes (and the choice of zirconium alloys).

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The construction components include the pressure tube, the joints, the thermal insulation and the liner tube. A brief account is given of the testing methods used and the performances of these various units is particular :

a) - the safety factors foreseen for the pressure tube, and the design and manufacture, taking account of tolerances, of the thickered ends necessary for fitting the tubes in place and designing the joints.

b) - the joints connecting the pressure tubes to the reactor tank, which are only accessible through the inside of the channel prolonging the pressure tube. These joints must not be a weak part in the construction. Two types have been developed : a rolled joint where the ends of the pressure tube are directly flanged onto the tank, and a welded joint using zircaloy-stainless steel transition pieces added to the ends of the pressure tube. All these joints are made by remote control and are removable.

c) - two solutions have been found to the problem of thermal insulation around a zirconium alloy liner tube. The neutron absorption equivalent is about 1,1 mm of Al, and the mean loss around 2 p. 100 of the thermal power of the reactor.

The methods proposed have proved practicable as a result of important research and developments on automatic remote control for all the operations which make up the sequences of mounting, demounting and reparing of the construction components. In particular the possibilities opened up by the new techniques of welding tubes from the inside have been extended to other problems connected with the assembling of a reactor.

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Abstract

PROBLEMS RELATED WITH THE POWER REGULATION OF REACTORS BY PHYSICO-CHEMICAL METHODS, AND THE BEHAVIOUR OF WATER AND HEAVY WATER IN NUCLEAR REACTORS

by MM. L. DOLLE - Service des Isotopes Stables
CEN-S, SACLAY (S. et O.) FRANCE
D. CONAN - Département des Piles Expérimentales
CEN-S, SACLAY (S. et O.) FRANCE
J. ROZENBERG - Service des Isotopes Stables
CEN-S, SACLAY (S. et O.) FRANCE
G. DIRIAN - Service des Isotopes Stables
CEN-S, SACLAY (S. et O.) FRANCE
E. ROTH - Service des Isotopes Stables,
CEN-S, SACLAY (S. et O.) FRANCE

(Commissariat à l'Energie Atomique, France)

The observation of the behaviour of water and heavy water in working reactors contributes to their safe running and provides information useful in studies relating to reactivity control techniques using soluble poisons.

The use of nuclear poisons dissolved in the water of a reactor leads to its chemical pollution. The conditions under which they can be used without causing the undesirable effects of this pollution are studied. Problems of analysis, although important, are not tackled in this paper.

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Behaviour of heavy water in working reactors.

1° - Isotopic pollution of heavy water : The rate at which this pollution occurs depends on the type of reactor and on certain characteristic incidents. The use of a reconcentration column is an efficient way of maintaining the heavy water isotopic concentration in a reactor which cannot be considered exempt from slow isotopic pollution.

2° - Heavy water leak detection : The instantaneous rates of small leaks can be measured, the leak localised, and the atmospheric contamination in the reactor building checked. Isotopic analysis for deuterium, or tritium determination, is carried out on samples.

3° - Chemical pollution and purification of heavy water : Chemical pollution of heavy water is one of the most complex problems in reactor chemistry. Corrosion of the materials which make up the core and the heavy water circuits varies extensively with the state of purity of the heavy water, as may be appreciated from the performances of the purification circuits and from direct measurements. From the knowledge acquired it has been possible to work out standards of purity which, if observed, will guarantee satisfactory running of the reactor.

4° - Radiolytic decomposition of heavy water : A better knowledge of its quantitative aspects in reactors is necessary in order to foresee the amounts of explosive gas mixture given off in power reactors. The radiolysis rate develops with the chemical purity of the water and the instantaneous power of the reactor. Experience of the CEA heavy water reactors and a systematic study of the radiolytic decomposition of water in the core of swimming-pool reactors are described.

Setting up of reactivity control by physico-chemical methods

1° - Reactivity control by homogeneous poisoning of the reactor : A comparison of the evolution of xenon poisoning with the residual antireactivity of the poison in solution during its nuclear consumption establishes the pro-

gramme which must govern the variation in its concentration if the exact compensation is to be produced.

The behaviour of the poison towards the reactor materials under the particular operational conditions must be taken into account.

2° - Radiolytic decomposition of water in the reactors in the presence of soluble poisons : A study of the effect of certain chemically inert salts, present in small concentrations in the water, on its radiolytic decomposition rate, has led to some new results which are discussed.

3° - The choice of a soluble poison is justified on the basis of the above results.

4° - Reactivity control by the use of a gaseous absorbant : The use of a "gas control rod" circuit for compensation purposes, in place of solid control rods is described.

The use of soluble poisons in the moderator to compensate the xenon effect, and of a gaseous absorbant in a circuit known as a "gas control rod", form original aspects of the reactivity control in the reactor EL 4.

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Abstract

Fe Al 40, A NEW CANNING MATERIAL FOR REACTORS USING REFRACTORY FUELS

by MM. G. SAINFORT
G. CABANE
M. SALESSE

(Commissariat à l'Energie Atomique, France)

Fe Al 40, owing to its high aluminium content, is more suitable than stainless stells for nuclear applications ; it has two advantages : its nuclear cross section is half that of 18-10 stainless steels, and its compatibility with fuel elements and heat extracting fluids is exceptionnally good.

Ferrous alloys with more than 16 per cent by weight of aluminium are reputed to be brittle because of their ordered lattice. But actually, most of the brittleness of these alloys is due to the presence of intergranular precipitates.

The vacuum casting of pure iron and aluminium, together with additions of scavenging elements, gives a very clean alloy with sufficiently reduced brittleness at high temperatures as to allow transformation with a very good yield.

Studies of smelting and transformation have enabled the optimum composition and the best industrial fabrication conditions to be established.

The mechanical properties of extruded or rolled products are dependent on the ferritic ordered structure of the alloys prepared as follows :

Extension at room temperature between 8 and 11 p. 100

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Continuous increase of the elongation at rupture with increase of temperature so that at 800°C it exceeds 100 p. 100,

Yield strength stable at 30 kg/mm^2 from 20 to 550°C

Progressive decrease of impact strength as the temperature increases

Creep strength at 650-700°C in the region of 10 kg/mm^2 with a very high elongation because of the appearance of a fine grain recrystallization during the test.

Welding of Fe Al 40 is facilitated by the fact that this alloy remains ferritic up to its melting point (1350°C) ; however it is sensitive to hot short cracking and grain swalling. Several welding processes have been successfully applied to this alloy, under conditions where the superficial alumina layer was removed.

This very impervious oxide skin gives Fe Al 40 a unique resistance to corrosion :

- in carbon dioxide at 700°C, 60 atm, the weight gain after 3000 h is about one third of that of a 18-10 niobium - stabilized stainless steel
- in water, either de oxygenated or saturated with oxygen at 25°C, the weight gain is one hundred time smaller than that of mild steel, after a 3 month test
- in water vapour at 500°C, also after 3 monts, only the growth of impervious and very adherent oxide skins is observed
- in sodium up to 1000 h at 700°C, the behaviour is at least as good as that of stainless steel.

No diffusion reaction has been detected either with uranium dioxide up to 800°C, or with uranium carbide up to 700°C.

Extension specimens and thin walled cans are subjected to high flux irradiations between 20 and 700°C ; preliminary results will be given.

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Abstract

WHY THE SEARCH FOR URANIUM SHOULD CONTINUE

by MM. MABILE
GANGLOFF
SORNEIN

(Commissariat à l'Energie Atomique, France)

At the 1958 Geneva conference a report was given of the hopes placed in French uranium resources, and the work in hand on projects for mines and mills. Even though the French ground continued to confirm its promises, production has been limited because of the risk of massive over-production. Moreover worlwide production surpluses remove any immediate worries about supplies.

1. - However, the French uranium potential, at the rate of exploitation which can be reasonably foreseen, seems to be limited, promising only a few decades of production.

2. - This situation is more serious still for Europe as a whole, at the very time when the rapid expansion of atomic energy is beginning.

3. - On a world wide basis it may be anticipated that after the closing down of mines, an equilibrium between production and requirments will be reached in ten years, the increase in production being then easily achieved by the reopening of temporarily closed mines.

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4. - However, by 1980 the requirments will exceed the maximum production capacity laid down in 1960-1961.

5. - Finaliy, towards the end of the century, there will be a discrepancy between the uranium reserves then necessary, and those which have been discovered so far in the world, at the cost of the considerable efforts made during the years 1948-1958.

No time should be lost in an effort to discover new resources :

- either in countries not yet prospected
- or by the use of more sensitive methods.

Failing this, the uranium marhet, badly shaken by present over-production, will suffer new crises unless the mining industry is well prepared in advance for a rapid exponential expansion after 1975-1980.

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Abstract

NEUTRON THERMALIZATION AND SPECTRA

by MM. M. CADILHAC
J. L. SOULE
O. TRETIAKOFF

(Commissariat à l'Energie Atomique, France)

The investigation of the neutron spectra in thermal reactors is taking an increasing importance due to the role played in them by Plutonium. Whatever the absorption law, it has been remarked that the scattering law in the moderator affects the spectrum only through certain overall properties. It would thus seem possible to develop a simplified representation of this effect which would lead to a clear understanding of the phenomena, reducing at the same time the volume of numerical calculations required.

The "synthetic" model employed by the authors presents the advantage of reducing the determination of the spectra in an homogeneous medium to the resolution of a second order differential equation, like the Wigner-Wilkins model (monoatomic gaseous hydrogen) and the "generalized heavy gas" model of J. Horowitz, which, incidentally, are both special cases. The model is, on the other hand, sufficiently general to allow a correct treatment of the situations met with in practice and in particular the important case where the presence of Plutonium introduces absorption resonances at low energy. Actually, the che-

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mical or crystalline bonds of the moderator are introduced into the proposed model through two energy functions. These functions have been adjusted for the usual moderators (Graphite, heavy water, light water) by means of known theoretical scattering laws.

In a heterogeneous medium, the most important factor is the mean spectrum in the fuel of one cell, the knowledge of which is allowed by a generalization of the Amouyal-Benoist-Horowitz method. The proposed model lends itself particularly well to such calculations and also allows the effects of rethermalization (for instance when the cooling system and the moderator are at different temperatures) to be treated.

Finally, some examples are given of practical applications :
codes for spectra and effective cross sections computations (editing of tables), codes for the treatment of neutron balance in a lattice or for the interpretation of experiments (spectral indices critical experiments on plutonium, diffusion parameters).

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Abstract

PROGRESS IN NEUTRON THEORY FOR THERMAL REACTORS

by R. NAUDET, A. AMOUYAL et P. BENOIST

(Commissariat à l'Energie Atomique, France)

A number of new theories and calculation methods for thermal neutron lattices are reported in this paper. Recent achievements in thermalization theory represent the main improvement in reactor physics : this work is described in a separate paper. But there are many other problems, and the target is eventually to obtain a better understanding of the overall neutron balance.

Among the topics the following ones are discussed :

- Calculation of the thermal utilization factor in a cell made up of a number of concentric media
- Relation between the migration area and the distribution in space and energy of the neutron absorptions (cut-off between fast and thermal leakage).
- Determination of the diffusion coefficients in a lattice containing cavities.
- Spectrum of slowing down neutrons and calculation of the resonance escape probability.
- Specific problems : applications of heterogeneous methods. Resolution of the transport equation in the double Pn code. Fine structure inside fuel bundles, etc.

In these various fields important refinements have been achieved and are already used in practical lattice calculations.

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Abstract

GRAPHITE REACTOR PHYSICS

by MM. BACHER - Département des Etudes de Piles - C. E. A.
CAILLY - Département des Etudes de Piles - C. E. A.
COGNE - Département des Etudes de Piles - C. E. A.
MEYER-HEINE - Département des Etudes de Piles - C. E. A.
MINNARD - Département d'Etudes Théoriques - EDF
NOC - Département d'Etudes Nucléaires - EDF
TAICLET - Département des Etudes de Piles - C. E. A.
TESTE DU BAILLER - Département des Etudes de Piles - C. E. A.

(Commissariat à l'Energie Atomique, France)

The study of graphite-natural uranium power reactor physics, undertaken ten years ago when the Marcoule piles were built, has continued to keep in step with the development of this type of pile.

From 1960 onwards the critical facility MARIUS has been available for a systematic study of the properties of lattices as a function of their pitch, of fuel geometry and of the diameter of cooling channels. This study has covered a very wide field : lattice pitch varying from 19 to 38 cm, uranium rods and tubes of cross - sections from 6 to 35 cm^2, channels with diameters between 70 and 140 mm. The lattice calculation methods could thus be checked and where necessary adapted.

The running of the Marcoule piles and the experiments carried out

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on them during the last few years have supplied valuable information on the overall evolution of the neutronic properties of the fuel as a function of irradiation. More detailed experiments have also been performed in Marius with plutonium-containing fuels (irradiated or synthetic fuels), and will be undertaken at the beginning of 1965 at high temperature in the critical facility CESAR, which is just being completed at Cadarache.

Spent fuel analyses complement these results and help in their interpretation. The thermalisation and spectra theories developed in France can thus be verified over the whole valid temperature range.

The efficiency of control rods as a function of their dimensions, the materials of which they are made and the lattices surrounding them has been measured in Marius, and the results compared with calculation on the one hand and with the measurements carried out in EDF 1 on the other. Studies on the control proper of graphite piles were concerned essentially with the risks of spatial instability and the means of detecting and controlling them, and with flux distortions caused by the control rods.

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Abstract

HEAVY WATER REACTORS PHYSICS

by Y. GIRARD, P. LOURME and R. NAUDET

with the collaboration of M. CHABRILLAC, G. LEDANOIS, G. MOUGEY, P.F. PALMEDO and PAVRET de la ROCHEFORDIERE

(Commissariat à l'Energie Atomique, France)

An important research programme on heavy water reactor physics has been carried out in France for quite a few years. The decision to build the EL 4 prototype and so to choose the heavy water gas cooled type has renewed the interest in this programme and at the same time given to it a more specific orientation. A summary of the results gained in this field is presented in this paper.

In the first part are described the experimental investigations, most of them were carried out in the criticality facility Aquilon II. The experiments are grouped in four parts :

- Systematic studies of lattices. Buckling measurements.
- Specific studies of gas-cooled lattices.
- Fine structure, spectral indices measurements etc ...
- Measurements on lattices or samples containing Uranium of various enrichment or Plutonium.

The second part is devoted to a summary of the theoretical studies. The whole results have allowed an improvement of the calculation methods, have led to a better understanding of the neutron balance in lattices, and have permitted the establishment of a set of formula to predict not only the clean fuel conditions but also the evolution of the nuclear properties with irradiation. Some specific studies on power reactor are quoted.

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Abstract

BURN UP PHYSICS

by O. TRETIAKOFF

(Commissariat à l'Energie Atomique, France)

The present communication is devoted to a body of theoretical and experimental work carried out at the C.E.A. with the aim of adding to the current knowledge on the evolution of the reactivity (during fuel irradiation) in natural or slightly enriched Uranium reactors.

The difficulties of performing direct experiments on large amounts of irradiated fuels are reviewed - especially in operating power reactors - and the necessity is underlined for fundamental research in two directions :

on one hand, the change in the composition of the fuels (chains of heavy nuclei, fission products),

and on the other hand the effect of changes in composition on the neutron balance.

Before presenting three types of experiments which have been carried out, the importance of the problems associated with the neutron spectra is stressed and the practical methods used for the calculations are briefly described.

The systematic irradiation of several types of fuel, followed by their chemical and isotopic analysis has been going on for several years. An outline

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of the experimental programme is given with a description of the methods employed : α, β, γ chain for the preparation of samples determination of the plutonium content by coulometry and double isotopic dilution, separation of Boron used in some cases for the measurement of integrated neutron densities. The interpretation of the measurements is discussed with some examples.

A second and more recent series of experiments deals with the investigation of lattices, using "synthetic" fuels (Uranium-Plutonium alloys) as compared to slightly depleted or enriched Uranium. Various experiments are considered on heavy water and on cold graphite, then on graphite heated up to 500°C. Some results already obtained are listed. These experiments, requiring nearly a metric ton of each type of fuel cannot be pursued in a systematic manner. This is why is developed since several years a method of differential measurement by oscillation, which requires samples of the order of several kilogrammes only. The relationships between these measurements and the investigations of lattices are discussed, and an outline is given of the way of carrying out the systematic study of fuels of various compositions. The method has been successfully applied to the systematic study of irradiated fuels (analysed independently by the methods mentioned above) thus giving the possibility of measuring "in situ" the absorption of fission products.

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Abstract

ESSOR, TEST REACTOR FOR THE HEAVY WATER PROJECTS DEVELOPED BY EURATOM

(Commissariat à l'Energie Atomique, France)

Essor is a trial reactor designed for specific tests on the heavy water reactor. Series studied by Euratom.

Its first job is connected with the Orgel project, where it will be used to test the channels (fuel elements and pressure tubes) and the behaviour of organic liquids as cooling fluids.

The reactor, heavy water-moderated, can test twelve channels grouped in a central zone, surrounded by a ring of booster fuel elements.

These booster elements consist of highly enriched uranium alloy fuel plates, cooled by forced circulation of heavy water.

The maximum thermal neutron flux in the central zone can reach 2.10^{14} n cm^{-2} s^{-1}

The experimental channels can be supplied by five cooling loops. In the first set-up three organic liquid loops will be installed.

That workshops adjoining the reactor enable the fuel elements and the pressure tubes to be examined and dismantled after irradiation. Research on a neutron distribution very similar to that of an Orgel type reactor, the means to carry on several simultaneous tests on the fuel elements and pressure tubes under conditions similar to real ones, a proportionate system for treating information, and numerous experimental facilities will make of Essor an indispensable techno-

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logical research tool in the development of the Orgel project.

The general characteristics of the reactor are as follows :

Power of the "Orgel zone"	13 MWth
Power of the "booster zone"	26 MWth

The 12 Orgel elements in a square lattice with 25.4 cm spacing

The 16 booster elements in a ring of 59 cm radius

Height of the elements 150 cm.

The Orgel zone is designed for 0.76 tons of uranium carbide.

The booster zone will take 5.9 kg of U-235.

Specific power : Orgel zone : 25 MW/t (max), 17 MW/t (average)

booster zone : 65 kW/g U-235 (max), 5.1 kW/g (average)

Maximum power per channel : Orgel : 1.12 MW

booster : 1.83 MW

Irradiation dose foreseen : 5000 MW d/T.

Although the Orgel fuel element is designed in the form of a cluster of 7 uranium carbide pencils canned with S.A.P., the use of other fuel elements is planned (oxide, metal, etc).

Construction of the reactor, begun in August 1963, continues actively and a divergence should be possible towards the end of 1966.

This is an example of international cooperation within the framework of Euratom ; the G.A.A.A. and Interatom Group having undertaken all the studies, has been assigned the industrial architecture of this unit in association with Montecatini.

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Abstract

STUDY OF THE DYNAMIC BEHAVIOUR OF THE REACTOR "RAPSODIE"

by R. ABDON - Département de Recherche Physique
Commissariat à l'Energie Atomique - France -

C. CAILLET - Département d'Electronique
Commissariat à l'Energie Atomique - France -

M. CHAIGNE - Département d'Electronique
Commissariat à l'Energie Atomique - France -

J. LADET - Département d'Electronique
Commissariat à l'Energie Atomique - France -

I. PUIG - Département de Recherche Physique
Commissariat à l'Energie Atomique - France -

CP. ZALESKI - Département de Recherche Physique
Commissariat à l'Energie Atomique - France -

It is known that fast neutron breeder reactors pose new problems in the fields of mechanics and heat technology, as well as in respect to their dynamic behaviour and safety.

The reactor RAPSODIE has been the object of numerous dynamic studies on analogue and digital computors for two types of fuel (metal and oxide).

After elaborating the mathematical models representing the whole of the installation (reactor block and cooling circuit) from the point of view of neutrons as well as thermodynamics, the analogue diagrams and digital codes required for simulating incidents, control and stability of the reactor were established.

Considerable effort was made to obtain a greater precision than in the usual methods, either by using smaller intervals, or by the use of formulations

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closer to the real system, or even analytically soluble.

The study of incidents was carried out on analog computer for the whole system, and digital computer for the reactor block alone or the installation functioning with a single thermal circuit.

A special complementary programme - which so far as we know, was used for the first time here for sodium reactors - which takes into account the behaviour of the reactor beyond the boiling point of sodium was refined on the basis of some simplifying assumptions.

The investigation of incidents has served to determine :

- The intrinsic safety of the reactor, i.e. its behaviour under accident conditions without the intervention of the safety systems ;

- The behaviour of the installation under normal accident conditions, i.e. with the safety systems functioning, and the definition and organisation of the latter ;

- The comparison of the two versions investigated (metal or oxide fuel) for the choice of the first loading of the reactor ;

- The determination of thermal shocks (amplitudes and gradients suffered by the structures during probable incidents) to be used in the test on the full-scale model of the reactor.

The stability tests were carried out digitally. The investigation of the control, carried out on analog computer, served to determine the different possible means of starting and changing the conditions of the reactor as well as its automatic control.

The calculations were examined in the totality by the construction of a training simulator composed of a board similar to the control board of the reactor, all of whose commands (reactivity and flows) work on an analogue computer which resolves in the real time the dynamic equations of the reactor and which reproduces simultaneously all the parameters representing the state of the installation (power, period, temperatures, etc .) in the case of various incidents as well as under normal conditions of functioning.

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Abstract

SOME PARTICULAR ASPECTS OF CONTROL IN NUCLEAR POWER REACTORS

by C. E. A. and E. D. F.
presented by J. FURET
Ingénieur D. E. /S. A. E. R. - C. E. N. SACLAY - FRANCE
(Commissariat à l'Energie Atomique, France)

There are still many problems in the field of measurement and control of neutron flux. The present studies in connexion with high flux reactors contribute to the solution of these problems which concern specialists in reactor control.

The present state of this investigation and the results of different studies carried out in France by the C. E. A. and the E. D. F. are pointed out :

A - In the nuclear instrumentation field, work is at present devoted to the technologies used to develop detectors and cables, which have to work at high temperature and in a high γ background ; fast electronic techniques are applied to fission counters to measure low neutron fluxes in a high γ background (10^6 Rh).

B - In the control and safety field, there is a real need for studies on the behaviour of reactors in the subcritical state. This increases the margin of security during restarts when poison effects must be overcome.

The perturbations due to control rod movements necessitate a new

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organisation of power level safety and control assemblies, in connexion with thermal or activation measurements.

Two methods of fast start-up are described. They are related to the fission rate measurement as a function of time. This is done either continuously by a constant and high reactivity change, or step by step.

The application of automatic techniques to detector motion seems to give the answer to control and safety in normal start-up.

C - The scope of these studies covers the methods used for the control of E. D. F. 3, which are described.

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Abstract

THE SAFETY OF GRAPHITE-GAS SEQUENCE REACTORS

by MM. FAYS[+] - LAURENT[++] - DE VATHAIRE[+++] -

Considerations of safety play an important part in the development of graphite-gas reactors, which are at present represented by the MARCOULE AND CHINON reactors, for which E.D.F.4 and E.D.F.5 reactors constitute the subsequent stages.

Safety comes into play during the design and construction stages of the structures, in particular involving an examination of the problems of building the containment vessel around the core and the primary cooling circuit (steel or prestressed concrete, external or built-in heat exchangers).

While operating, safety is particulary dependent on well-chosen arrangement of the instrumentation in the core, on the satisfactory operation of automatic devices and safety sequences, and on any waste. These various points are examined in turn. Safety requirements also give rise to upper power ceilings either in steady-state or transient regimes.

The principal incidents liable to occur are reviewed, and the theoretical and experimental means of studying them, and the steps taken to minimise their consequences are examined.

+ Head of Safety Division at the E.D.F. General Nuclear Design Department
++ Head of Reactor Division at the E.D.F. Number 1 Nuclear Region
+++ Head of the C.E.A. Working Party on Reactor Safety

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Abstract

SAFETY IDEAS IN FRANCE AND THE INFLUENCE OF SAFETY DEMANDS ON REACTOR DESIGN

by MM. de VATHAIRE - Département des Etudes de Piles - CEA Saclay
DEVRON - Département de Recherche Physique - CEA Saclay
MEUNIER - Département des Etudes de Piles - CEA Saclay
PASCOUET - Pyrotechnie de Toulon
VERNIER - Service des Transferts Thermiques - CEA Grenoble

(Commissariat à l'Energie Atomique, France)

This paper reviews the experience acquired in France on the question of reactor safety.

Since a special paper is being presented on reactors of the graphite-gas type, the safety of the other types studied in France is discussed here :

- heavy water-gas reactors,
- fast neutron reactors,
- water research reactors of the "swimming-pool" and "tank" types.

The safety rules peculiar to the different types are explained, with emphasis on their influence on the reactor designs and on the power limits they impose.

The corresponding safety studies are presented, particular stress being placed on the original work developed in these fields. Special mention is made of the experimental systems constructed for these studies : the reactor CABRI, pile loop for depressurization tests, loops outside the pile, mock-ups etc...

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Abstract

REVIEW OF SAFETY PRECAUTIONS DERIVED FROM SIX-YEAR EXPERIENCE OF A SPENT FUEL REPROCESSING PLANT

(Commissariat à l'Energie Atomique, France)

In a center for the processing of irradiated fuels, as in any other factory, continuity of production can only be achieved through constant effort to maintain the standard of safety. The authors examine the safety elements inherent in the design of the installations, from the point of view of both radioactive and conventional protection.

It is clear that in the nuclear industry it is not sufficient merely to adopt the usual precautions against the risks of explosion, fire, etc...; but great attention must be paid to the problems arising from the confinement of materials, radioactivity and the risks of criticality.

The general principles of the technical prevention of radioactive risks are now well known.

They must be applied as soon as the plans are drawn up for the installations, by taking into account :

- the design and construction of the installations with respect to the risks mentioned above
- the division of the plant into zones
- the circulation of personnel and material
- ventilation and purification of the air

The technical safety factors however, are not sufficient in themselves. However good the material and the degree of automation achieved, the smooth running of a factory depends on the man responsible for supervision and repair

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work or who takes initiatives. Technical safety is never definitive and must be continually maintained. The collaboration of the workers is necessary to its efficiency. Psychological measures must thus be taken as an addition to the technical precautions. This psychological prevention is based on :

- education of the personnel
- organization of the work and definition of the safety rules
- checking on obediance to the rules and behaviour of the personnel
- constant reminders of the necessity for the prevention of radioactive risks.

Six years operation of the irradiated fuel processing plant at Marcoule have shown the importance of close collaboration between workers and technique. The authors analyse the results obtained from an organization of the work aimed at eliminating the causes of incidents. They investigate the following points in particular :

- frequency of entry into forbidden zones during normal functioning
- the distribution of exceptional tasks
- the prevention of criticality risks
- zone contamination incidents
- the volume and nature of decontaminated material and eliminated residues
- the evolution and distribution of external irradiation doses received by the workers
- the frequency of cases of body contamination
- internal contamination fixed in the organisms of workers.

Conclusions are drawn from the different incidents which have occured during the years of operating the plant.